# Pressurized Thermal Shock, PTS

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**Abstract.** Pressurized Thermal Shock (PTS) refers to a condition that challenges the integrity of the reactor pressure vessel. The root cause of this problem is the radiation embrittlement of the reactor vessel. This embrittlement leads to an increase in the reference temperature for nil ductility transition (RTNDT). RTNDT can increase to the point where the reactor vessel material can loose fracture toughness during overcooling events. The analysis of the risk of having a PTS for a specific plant is a multi-disciplinary problem involving probabilistic risk analysis (PRA), thermal-hydraulic analysis, and ultimately a structural and fracture analysis of the vessel wall. The PRA effort involves the postulation of overcooling events and ultimately leads to an integrated risk analysis. The thermal-hydraulic effort involves the difficult task of predicting the system behavior during a postulated overcooling scenario with a special emphasis on predicting the thermal and mechanic loadings on the reactor pressure vessel wall. The structural and fracture analysis of the reactor vessel wall relies on the thermal-hydraulic conditions as boundary conditions. The US experience has indicated that medium and large diameter primary system breaks dominate the risk of PTS along with scenarios that involve a stuck open valve (and associated system cooldown) that recloses resulting in system re-pressurization while the vessel wall is cool.

## 1. INTRODUCTION

A brief overview of issues related to pressurized thermal shock (PTS) is provided in order to facilitate an understanding of the complexity that one is faced with when attempting to address PTS safety concerns at nuclear power plants. PTS analysis requires a multi-disciplinary approach due to the wide range of issues that must be considered. These disciplines include probabilistic risk analysis (PRA), materials research, thermal-hydraulics, structural mechanics, fracture mechanics, and materials testing and inspection. A careful examination of PTS requires a team of experts with specialties in each of these subjects. This report covers these topics only superficially to provide a broad overview of the challenges faced in a PTS examination.

PTS refers to a condition that challenges the integrity of the reactor pressure vessel (RPV). The integrity of this primary component in a nuclear power plant is vital which explains why the PTS events receive the attention in safety analyses. Three conditions are typically considered as prerequisites that lead to a PTS safety concern. These include neutron embrittlement of the RPV, some type of existing flaw, and finally a rapid drop in temperature in the system (overcooling event).

In the United States (US), the overcooling event at Rancho Seco in 1978 raised concerns about the possibility of PTS. In this event, a failed power supply caused the loss of some control room instruments. The plant's integrated control system (ICS) reduced main feedwater causing the reactor to trip on high pressure. A cooldown started when feedwater was setup on a steam generator by the ICS. The cooldown reduced system pressure which initiated the high-pressure injection (HPI) pumps along with feedwater to both SGs. Pressure climbed to 2000 psi. The operators, still trying to restore control room power, continued feedwater to the SGs and maintained primary system pressure with the HPI pumps. Later analyses of the event by the Nuclear Regulatory Commission indicated the vessel could have failed under these conditions if the RPV was more embrittled and if an appropriately sized flaw was present. Subsequent control system modifications along with improved operator training and procedures have significantly reduced the possibility of this type of event occurring again. However, other types of overcooling events have been postulated and PTS remains a safety concern for many nuclear safety analysts.

Basic overcooling events include both primary and secondary side loss of coolant accidents (LOCAs). The primary side events are generally classified by break size and location. Breaks larger than about 3.5 cm typically result in loop flow stagnation as the upper regions of the steam generators void and full loop natural circulation is halted. The injection of cold water (tank temperatures can range from 10o to 30o C) into the stagnated loops results in a significant cooldown. US plants typically inject into the cold leg and the cold flow quickly enters the downcomer and comes in contact with the RPV wall. Smaller breaks generally are associated with slower cooldown rates and a system that remains at a relatively higher pressure. Larger breaks accelerate the cooldown rate along with a significant drop in pressure. The larger breaks subject the vessel to the largest thermally induced shock. One particularly challenging event type is a stuck opened valve which re-closes after the temperature has had time to drop off. This accident produces both high pressure and low temperatures in the vessel wall.

Secondary side breaks, such as a main steam line break (MSLB), reduce the secondary side pressure and the water boils at low pressure in the steam generator. The steam generator, being an excellent heat exchanger, quickly cools down the primary side temperatures. Primary system temperatures can approach 100o C in the steam generator and natural circulation draws the cooler water into the downcomer where it cools the RPV wall.

The overall analysis of PTS is a multi-disciplined exercise that requires a team with a broad range of skills. Figure 1 illustrates a flow chart that depicts one approach to PTS analysis.

Plant specific details can play a significant role in a PTS evaluation. Besides the obvious plant specific nature of the vessel walls and embrittlement, operator procedures, control system set points, safety injection flow rates, injection geometry, and other system details can impact the severity of the PTS transient. The gathering of the plant specific information is the first step in a PTS analysis.

Another step in the PTS evaluation is a determination of potential overcooling events. This is typically carried out with a probabilistic risk analysis which determines a list of overcooling events. This process typically requires some type of broad screening criteria with support from thermal-hydraulic system codes to look for events with a significant probability of occurring along with the potential for overcooling. A typical analysis might result in thousands of individual scenarios which must be grouped into a manageable set of scenarios to facilitate the analysis. For instance, primary side breaks could be grouped into ranges of break sizes and a single representative scenario could be used for each group. Grouping is accomplished by looking at the PRA generated event trees with some preliminary thermal-hydraulic analysis to support grouping. Ultimately this becomes an iterative process. As more thermal-hydraulic analysis is completed, the PRA analysis is updated and the groups of scenarios are adjusted. Ultimately, a manageable number of overcooling events (groups) are identified for final thermal-hydraulic analysis.

The goal of the thermal-hydraulic analysis is the determination of the pressure, temperature, and heat transfer histories in the downcomer region that affect the RPV wall thermal and mechanical loading. System codes such as TRACE, RELAP5, ATHLET, or CATHARE are used to predict the overall system behaviour during the postulated events. If the system code models are insufficient to accurately predict the local downcomer conditions, due to some multi-dimensional effects for instance, the results can be augmented with additional models that can account for the parameters that govern the downcomer conditions.

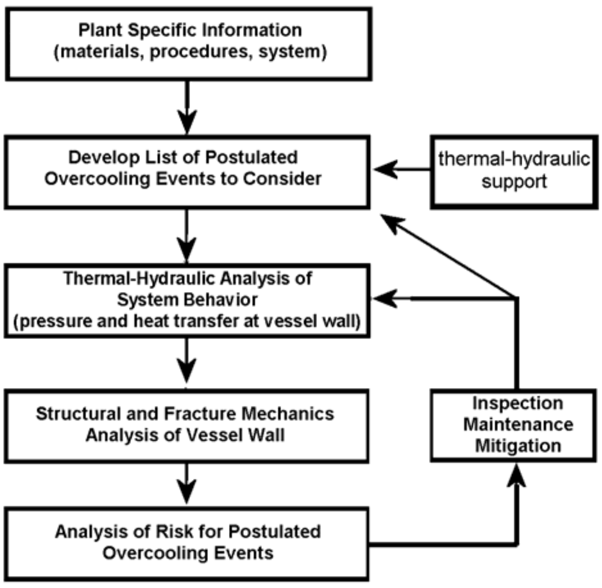



**Figure 1**: An Analysis Approach to PTS.

To determine if an overcooling event will result in failure of the RPV, a structural and fracture mechanic analysis of the vessel wall is completed to determine if the overcooling event can cause a through wall crack. These analyses use the thermal-hydraulic conditions in the downcomer region as boundary conditions.

Once the analysis of the vessel wall is completed for all scenarios, or groups, an integrated risk assessment for the plant can be completed. The risk dominant sequences can be determined along with root cause analyses which can lead to maintenance or mitigation strategies to reduce the probability of these events occurring or the severity of the event itself.

## 2. NEUTRON FLUENCE (THE ROOT CAUSE)

PTS is not a concern for new vessels which have high fracture toughness. It is only the older plants with specific material properties that are susceptible. High energy neutrons coming from the core are the primary cause of the RPV embrittlement. Neutron fluence (neutrons/$m^2$) is simply the integral of the neutron flux (neutrons/$m^2$ –s) over time.

Hardening (or embrittlement) starts at the nanometer level as the high energy neutrons are absorbed by the material causing lattice defects which cluster. Details are beyond the scope of this report. The important thing to know, however, is that the neutron fluence causes the material to loose fracture toughness and in addition causes a shift upwards in the nil ductility transition temperature (RTNDT). It is this shift in the RTNDT value that is the heart of the problem. Consider figure 2 which illustrates Charpy-V-Notch (CVN) results for material before and after irradiation. The CVN test is an old approach to measure the fracture toughness of a material that is still in use today. Improved techniques are available but the Charpy results can be used to clearly show the impact of neutron fluence.

The upper curve on Figure 2 represents the CVN energy curve for the unirradiated specimens. The test measures the energy required to break a specific specimen at a given temperature. If the material is ductile (tough), the energy required to break the specimen is high. As the material is cooled, it looses fracture toughness (becomes more brittle) as illustrated by the curves in Figure 2. The upper curve shows relatively high fracture toughness for temperatures greater than 20$^o$ C. Since RPV temperatures are not expected to drop to this level, this material will remain tough during an overcooling event.

The lower curve on Figure 2 illustrates the results of CVN testing for irradiated specimens (1.5 x 1019 neutrons/cm2 (>1MeV)). There is a 33 percent shift downward in the upper shelf of the fracture toughness and about an 82$^o$ C shift upward in the nil ductility transition temperature in this example. It is this shift to the right (to higher temperatures)

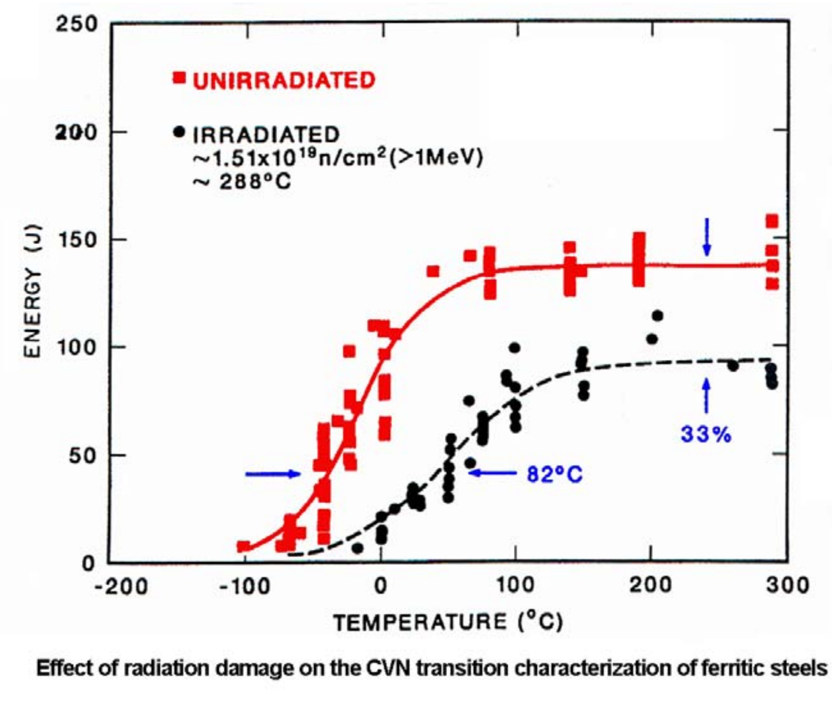



**Figure 2:** Courtesy of ORNL.

of the transition temperature that causes the greater concern. Primary system overcooling events result in safety injections with injection water temperatures around 20$^{o}$ C. In this example, if the irradiated vessel wall approached the injection temperatures, the fracture toughness of the material would be lost.

At the beginning of life, the vessels are designed to be tough. It is though time, and the associated neutron fluence associated with normal operation, that the toughness curve will shift as shown in Figure 2. Some older welds can be particularly susceptible to embrittlement. Trace impurities, like copper and phosphorous, were not carefully controlled in some early plant welds and this has led to increased concern for those welds.

## 3. THERMAL-HYDRAULICS

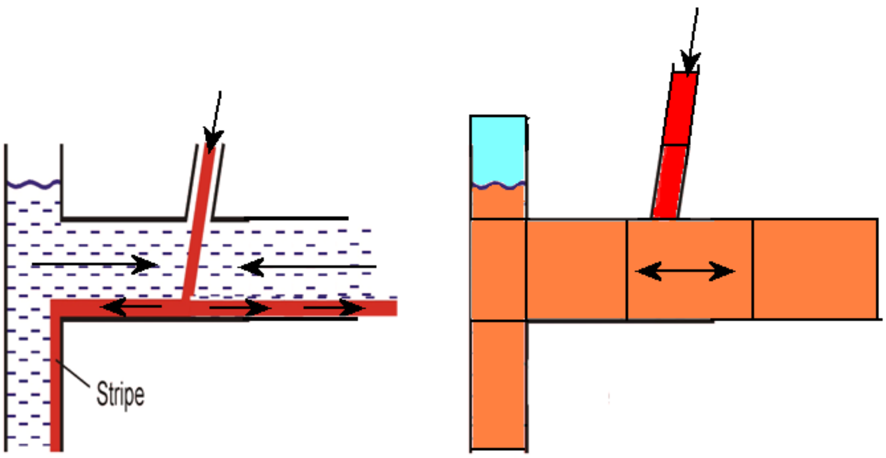
The goal of the thermal-hydraulic analysis is the prediction of the pressure, temperature, and heat transfer coefficient histories in the downcomer region. These conditions serve as thermal and mechanical boundary conditions for the RPV wall.

The prediction of overall plant response to the postulated overcooling scenario is a first priority. System codes such as CATHARE, TRACE, RELAP5, and ATHLET, are used for this purpose. These codes are widely used for the prediction of plant behavior under a variety of accident conditions.

For primary system breaks, plant specific behavior and details of the modeling and operator actions can be very important. The location, timing, flow rates, and temperatures of both accumulator and safety injection flows are important. Break size and location are also important. For instance, in a cold leg break, some of the safety injection flows could simply exit the break directly. Another feature found to be of some significance is the upper plenum bypass flows. This flow path allows hot fluid from the upper plenum to bypass into the upper downcomer. This provides hot water to mix with the cooler safety injection flows and helps to reduce the thermal shock.

The modeling of physical processes such as the timing of the interruption and resumption of loop flow natural circulation flows is important. The safety injection flows are mixed much more effectively when full loop flow natural circulation is underway. Under the conditions of loop flow stagnation, fluid mixing and/or condensation on safety injection flows is also of concern of the thermal-hydraulic modeler.

After the system code predictions are completed, it is common to consider the applicability of these predictions for local downcomer behavior. In some instances, the 1D modeling approach common to the system code models needs to be augmented. This is a general feature of system code modeling. In regions where 3D behavior or some other phenomena occurs which is outside of the normal assumptions of the system code model, additional modeling is completed to quantify the impact of the multi-dimensional or other effect. In PTS analyses, it is common for the analyst to apply regional mixing models , zonal models, CFD techniques, or some other specialized tool to predict the thermal-hydraulic behavior in the downcomer region during a PTS event. It is generally accepted that the system code predictions are used to provide boundary conditions for the detailed model in the region of the downcomer. Many detailed models include part of the loop seal region, the cold legs, the downcomer, and the lower plenum. In some cases, the specialized code or approach is coupled to the system code solution so that the detailed multi-dimensional behavior can provide feedback to the overall system response.




3A. Physical behaviour 3B. Predicted Behaviour

**Figure 3:** Behaviour of Fluid in Cold Leg during Safety Injection.

One example where the system code assumptions are not adequate is the case of safety injection into a stagnated cold leg full of water. Figure 3 illustrates a comparison of the expected physical behavior alongside the system code predicted behavior for a portion of the cold leg and the upper downcomer region.

Figure 3A illustrates a simple diagram of the expected behaviour. The relatively cold, and therefore heavy, safety injection (shown in red) enters from the top of the cold leg and accumulates on the bottom of the pipe where it flows horizontally in both directions (towards the downcomer and towards the steam generators). Some mixing occurs on this falling plume and the flows in the upper portion of the pipe represent the hot flow moving towards the injection point to make up for the hot fluid entrained by the incoming injection. The cold fluid at the bottom of the cold leg enters the downcomer as a cold stripe (or falling plume) which cools the downcomer wall in the vicinity of the stripe. Mixing occurs at the injection point, along the boundary of the horizontal counter current flows, in the transition of the flows into the downcomer, and ultimately as the cold plume decays into the warmer fluid in the downcomer.

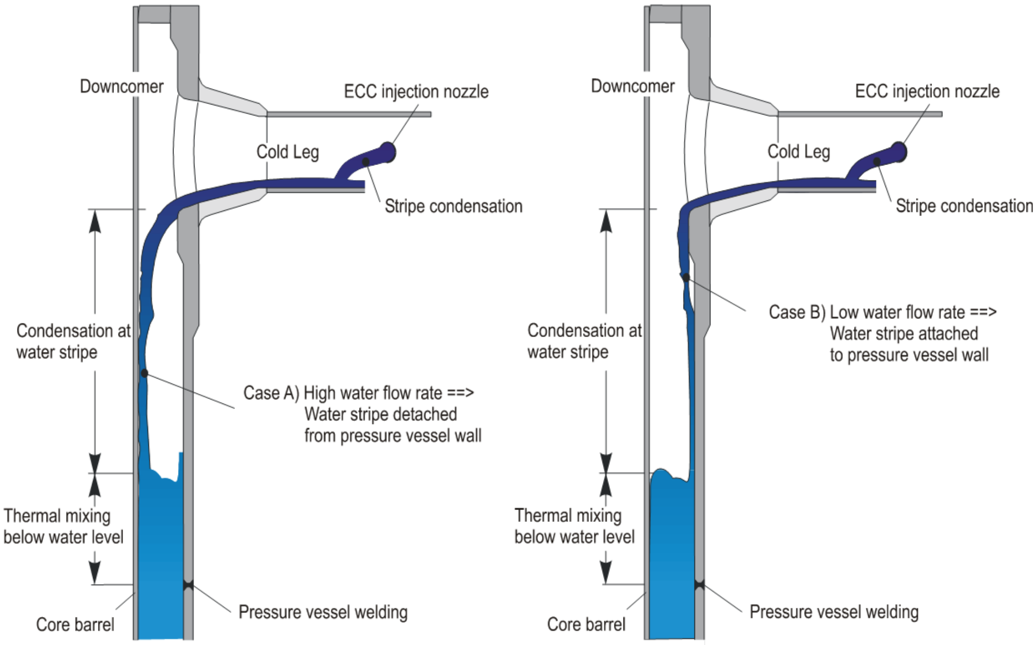
Figure 3B shows a simplified example of the system code prediction. Because of the 1D assumption in the cold leg, each cell has the same liquid velocity and temperature. Values can change from cell to cell but within each, the liquid is fully mixed. The counter current flow in the cold leg, shown in Figure 3A, is not predicted in the system code formulation. Downcomer flows are similarly limited. The cells are typically much larger than the width of the thermal stripe and this causes a rapid spreading of the cold flow. Many system codes use a 2D nodalization of the downcomer region (vertical and azimuthal cells). Six or eight azimuthal cells are common. The predicted width of the cold region in the downcomer under the cold leg is directly related to the downcomer nodalization. The turbulent diffusion and mixing of the cold negatively buoyant plume in the downcomer is not modelled. Mixing is expected to be overestimated by the numerical diffusion in the system code model. In addition, system code models tend to predict some unphysical oscillations in the cold leg flows under these conditions. It is clear to see why the system code formulation does not account for the expected physical behaviour. It is up to the analyst to determine the significance of these issues in light of the overall uncertainty that can be accepted and the potential significance of TH results.

The fluid-fluid mixing problem illustrated in figure 3 has been the subject of numerous experiments. These have included tests in the USA (CREARE-1/5 -1/2 , Purdue-1/2 , SAI/EPRI (SAI-1/1 ), Japan (-1/3 ), Finland (IVO-2/5 ), Belgium (UCL/TRACT-1/2 ), and Germany (HDR-1/1 , UPTF-1/1 ). These data have been used to validate models and correlations, regional mixing models, and CFD tools for use in the evaluation of PTS scenarios.

A more challenging condition which is receiving more attention in the past 10 years is the condition of safety injection into voided or partially voided cold legs. The voided cold leg with a reduced water level in the downcomer was the subject of an international comparative assessment in the late 1990's. Part of this study included liquid solid conditions similar to those discussed above. The most challenging thermal-hydraulic conditions, however, involved a reduced water level and safety injections into the steam environment. Figure 4 from Thicket-2004 (Hicken ) illustrates the injection flow conditions into a voided cold leg with a reduced water level in the downcomer. Figure 4A illustrates a case where the flow from the cold leg detaches and jumps to the inner wall of the downcomer. This behaviour is expected for mass flows greater than 10 kg/s (ref. E.F. Hicken, 2004). For mass flows less than 10 kg/s, the water stripe is expected to stay attached to the outer wall of the vessel as shown in Figure 4B. The steam in the cold leg and downcomer is expected to condense on the surface of the injection flows and water stripe. This type of direct contact condensation along with PTS has been identified by the EUROFASTNET project as a key safety related issue (see Lucas 2008). The condensation provides the main heating mechanism for the injected water. High condensation rates will warm up the injected fluid while a reduction in the condensation rate would allow the cold water to make its way into the downcomer at a lower temperature. Factors such as liquid surface area, heat transfer rates, splashing, droplet entrainment, turbulence levels within the liquid and gas, non-condensable gas concentrations, and the water stripe detachment mechanism are all important considerations for this type of modelling.

The issue of PTS and direct contact condensation modelling are being addressed by the NURESIM (SP2) Thermal-Hydraulics project . The focus of this work is on two-phase flow phenomena with an emphasis on modelling needs and areas for improvement. There is a great amount of work going on in the project (and worldwide) in the area of two-phase thermal-hydraulic analyses and the PTS issue will benefit from these developments. A recent paper (D. Lucas and D. Bestion 2007) which considers the case of the partially voided cold leg provides a good overview of the state of the art for this type of analysis. The results demonstrate a qualitative understanding of the individual phenomena of interest but highlight the need for further work in the area of combining the individual aspects into a robust and validated tool.

With the increasing speed of modern computers, CFD techniques are becoming more widely used and may provide the best tool for computing the thermal fluid mixing observed in Figure 3A. These approaches, however, still suffer from considerable user effect and the need for best practice guidelines continues. Challenges for the single phase




4A. mass flow > 10 kg/s 4B. mass flow < 10 kg/s

**Figure 4:** Safety Injection Flows into Voided Cold Leg (ref. E. F. Hicken, Thicket 2004).

CFD user include the turbulence modelling approach which must be able to adequately simulate the various mixing regimes which each have their own unique geometry and driving forces. In addition, the wall modelling approach is important. Mixed convection in the downcomer region is expected and typical CFD wall treatments do not account for this phenomenon.

All of the issues associated with CFD for the single phase PTS issue are compounded in the multi-phase problem by the relative immaturity of the multi-phase CFD techniques. For the near future, system analysis codes will still provide the overall system behaviour and experimental results will provide the best source of information on the details of the multi-phase behaviour related to PTS. Regional or zonal models, which require extensive experimental validation, can also play a role. The author is optimistic, given the efforts that can be observed world wide, that two-phase CFD techniques will continue to evolve and will play an ever increasing role in nuclear safety analyses including PTS.

### 4. PTS-ICAS 1996-1998

An International Comparative Assessment Study (ICAS) of PTS (NEA/CSNI/R(99)3) took place in the late 1990s. The ICAS included deterministic fracture mechanics, probabilistic fracture mechanics, and thermal-hydraulic mixing. This program was a follow on to the FALSIRE (focused mainly on fracture analysis) program completed earlier. The ICAS study added a thermal-hydraulic task. Results from this study were discussed during THICKET-2004 (Hicken E.F., Paper No. 29, Thicket 2004). Details of the ICAS will not be covered in as much detail in this report. Some of the conclusions taken from the report are copied below.

In the deterministic fracture mechanics group, the user effect was still an issue. There was even initial scatter in the thermal conduction solutions (surprising for a problem where the boundary and initial conditions are specified along with the material properties and dimensions). Where careful attention to detail was maintained, reasonable agreement was obtained in the stress analysis results. The authors found that consistent solutions required an adequate representation of the thermal and pressure transients, sufficient mesh density with quadratic elements recommended and correct material properties. These items seem to fall under the category of user guidelines and user experience. Further details are available in the summary report.

In the probabilistic fracture mechanics group, the calculated conditional probabilities of crack initiation have the largest scatter (a factor of 100). The conditional probability of vessel failure had a range of scatter from 20 to 50. Some solutions were obtained using the same computer code and still showed significant scatter based on some input selections. It would appear that the user effect is significant and it is noted that this is for a relatively well defined problem. One might expect significantly more scatter if each analyst had to collect information and make assumptions from scratch for a given nuclear power plant.

In the thermal-hydraulic task, correlation based approaches, system codes, and CFD tools were used to compute the thermal mixing and condensation rates under a variety of conditions. Large scatter in the results is seen in the cases where the water level is reduced (similar to Figure 4). Solutions with the lower temperatures seemed to underestimate the condensation while some solutions overestimated it. The correlation based approaches are considered the most appropriate. Some of these are based on similar data from the same facility. The heat transfer coefficient, with values ranging from 0 to 10 kW/m2-K, shows the most scatter.

This exercise highlighted some of the problems with thermal-hydraulic approaches. The correlation based approaches are appropriate, but only when closely tied to expensive experiments with similar geometry and flow conditions. The system codes sometimes suffer from simplified models that do not account for multi-dimensional nature of the phenomena taking place. These too are only as good as the assumptions on which they are based. The fundamentally based zonal models provide an efficient technique that can provide reasonable answers. The methods, like others, require significant validation with experimental results. The CFD tools, specifically for the two-phase problems, still need to be developed as discussed above. The results of the PTS ICAS are now 10 years old. Since then, the system codes have been steadily improved and offer more robust models for condensation effects. CFD models have also benefited from significant development. A major issue remaining to address is the user effect. Best practice guidelines are a good start but still appear to lack sufficient detail to ensure consistent results.

## 5. RECENT US EXPERIENCE

The US NRC has recently put forth a technical basis to revise the PTS screening criteria based on an updated risk-informed PTS analysis. The general approach to the resolution was a probabilistic risk assessment using RELAP5 to evaluate the overcooling events and the FAVOR probabilistic fracture mechanics code to estimate the probability of cracks penetrating the RPV wall.

Several basic conclusions could be reached from these studies. First, primary side breaks dominate the through wall crack frequency (TWCF). Secondary side breaks played a much smaller role. For the primary breaks, it was found that the large breaks dominated the TWCF. Another challenging event is a stuck open valve on the primary side which later recloses. This event cools down the RPV wall as the valve is opened and then the pressure recovers rapidly when the valve is closed. Figure 5 (Kirk, et al., NUREG 1806) shows a brief summary of the results.

The results from the US study found that the severity of the typical overcooling scenario was controlled by the initial cooling rate, the minimum temperature, and the pressure retained in the primary system. The transient severity is integrated with the transient likelihood (event frequency) to obtain the overall significance of the event.

RELAP5 was used in the US study to predict the downcomer temperatures and pressures. Some experimental work, CFD analysis, and regional mixing models were used to support the thermal-hydraulic evaluations. However, in the framework of the probabilistic approach used to determine the TWCF, the variations within the downcomer

**Factors contributing to the severity and risk-dominance of various transient classes**

| Transient Class | | Transient Severity: Cooling Rate | Transient Severity: Minimum Temperature | Transient Severity: Pressure | Transient Likelihood | TWCF Contribution |
|---|---|---|---|---|---|---|
| Primary Side Pipe Breaks | Large-Diameter | Fast | Low | Low | Low | Large |
| | Medium-Diameter | Moderate | Low | Low | Moderate | Large |
| | Small-Diameter | Slow | High | Moderate | High | ~0 |
| Stuck-Open Valves, Primary Side | Valve Recloses | Slow | Moderate | High | High | Large |
| | Valve Remains Open | Slow | Moderate | Low | High | ~0 |
| Main Steam Line Break | | Fast | Moderate | High | High | Small |
| Stuck-Open Valve(s), Secondary Side | | Moderate | High | High | High | ~0 |
| Feed-and- Bleed | | Slow | Low | Low | Low | ~0 |
| Steam Generator Tube Rupture | | Slow | High | Moderate | Low | ~0 |
| Mixed Primary & Secondary Initiators | | Slow | Mixed | | Very Low | ~0 |

| Color Key | | | |
|---|---|---|---|
| Color Key | Enhances TWCF Contribution | Intermediate | Diminishes TWCF Contribution |

**Figure 5:** Summary of Factors Leading to TWCF.

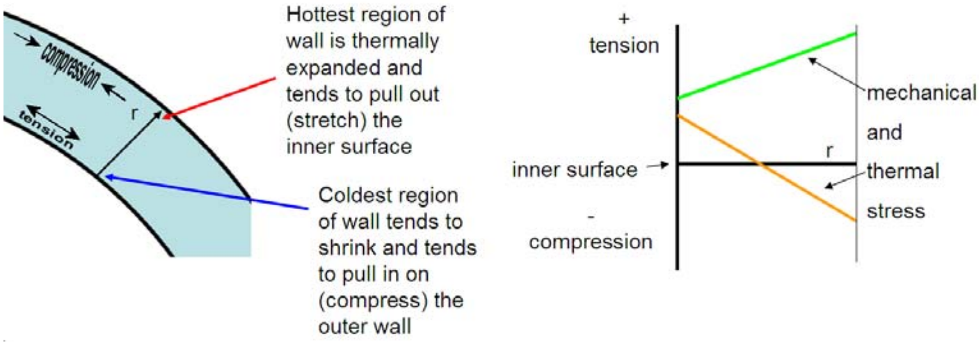



**Figure 6:** Hypothetical Wall and PTS Stresses.

temperatures and heat transfer were within the thermal-hydraulic uncertainty (for a given transient group) and therefore proved to be un-important in the overall evaluation. A large contributor to the thermal-hydraulic uncertainty was the boundary conditions assumed in the RELAP5 model.

It is important first to understand the basic concept of the PTS. The “pressurized” component of PTS refers to the pressure loading on the vessel wall. Hoop stress is the primary component of this loading along with the axial stresses. The “thermal” component refers to the loading caused by the overcooling of the inner wall of the vessel. The inner portion of the thick vessel wall is cooled and contracts while the outer regions of the wall remains warmer. If one imagines the vessel wall as two concentric cylinders, the inner cylinder is cooled and contracts. This tends to compress (pull in on) the outer cylinder. Meanwhile, the outer cylinder tends to pull out on (tensile stress) the inner cylinder. Figure 6 illustrates a hypothetical wall under a pressurized thermal shock with a plot of the thermal and mechanical stresses. The graph shows the mechanical and thermal stresses. Mechanic stresses are always positive. The thermal stress is high at the inner wall with the material in tension. The stress drops off and reaches zero before going negative (compressive stress) in the outer regions of the wall.

## 6. STRUCTURAL AND FRACTURE MECHANICS

The PTS is typically computed using standard 1D or 3D finite element methods. Total stress can be computed as a function of time using the thermal and mechanical boundary conditions from the thermal-hydraulic predictions as boundary conditions on the inner wall. Stresses in the unflawed RPV wall are typically below the yield strength of the material. The issue is non-ductile failure associated with the loss in fracture toughness (radiation embrittlement).

The science of fracture mechanics is used to determine whether a flaw in the vessel will become a through wall crack under the given loading. There are different approaches to the fracture analyses and only a simple description is provided here. The material has a fracture toughness value, indicated by K1C, which is a measure of the materials resistance to brittle fracture starting from flaws in the material. K1C refers to the toughness of the material in mode 1 fracture and K1C is a function of temperature. A mode 1 fracture example is depicted in Figure 7. This mode tends to open the flaw up directly.

K1C is the measure of the toughness that resists the stress intensity factor, K1. If K1 is greater than K1C, then the flaw will propagate. Using the principle of superposition, a stress intensity factor (K1) can be computed as a function of the stress field, the flaw geometry, and the flaw orientation. Inner surface breaking flaws have the

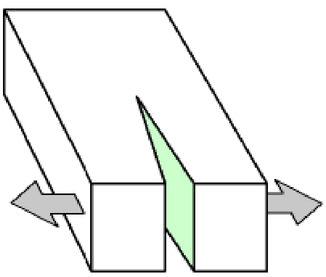


**Figure 7:** Mode I Fracture.

highest values of K1 for a given stress field. Both K1 and K1C vary during the overcooling transient. For a given flaw length and orientation, K1 varies due to the variations in the stress field during the transient. K1C varies with the temperature of the material. During the scenario, K1 and K1C are compared to see if the flaw propagates. The geometry of the vessel and the loadings associated with this geometry result in axial flaws having higher K1 values than circumferential flaws.

Crack arrest is also typically considered in the fracture mechanics models. The value of K1a is used to determine if a crack stops. If K1 is less than K1a, the crack will stop propagating. Many flaws will initiate (K1 > K1C) and then arrest (K1 <K1a) deeper in the vessel wall. This is understandable in light of the thermal stress profile shown in Figure 6. It is also noted that circumferential flaws are more likely to arrest. For axially oriented flaws, the stress will increase as the flaw grows.

This discussion only hints at the complexities of the structural and fracture mechanics analyses which are beyond the scope of this report. Topics such as weld residual stresses, pre-stressing, and stress discontinuities at the cladding to base junction are examples of issues that are also considered in a detailed analyses.

Recent US experience found that larger breaks are risk dominant and that under these loading conditions the thermal stress is higher than the pressure induced stress. For re-pressurization cases, the pressure induced stress is critical. Destructive evaluations completed in support of US efforts indicate that flaw density is highest in the welds. Fusion line defects are most common. This means that axial welds typically have axial flaws and circumferential welds typically have circumferential flaws. For the reasons noted above, axial flaws dominate the through wall crack frequency.

## 7. MAINTENANCE AND INSPECTION

Maintenance and inspection activities are common for nuclear power plants for more than just PTS issues. Some of the activities which support issues related to PTS are noted below. As noted earlier, the primary issue for PTS is the embrittlement of the vessel. Surveillance capsules are used to maintain specimens within the vessel which can be pulled from the capsule and sent out for testing on routine intervals. These can be positioned in areas of increased neutron flux to increase the neutron fluence and therefore get an earlier prediction of the vessel embrittlement. Fluence monitoring is another activity that is common and shielding is an option to delay the embrittlement at key locations within the vessel. Annealing the vessel is one method of reversing the vessel embrittlement.

Detecting flaws in the vessel is another goal of inspection activities. As part of the in service inspections, it is common to have a visual inspection of the internal vessel surfaces as well as ultrasonic inspections of key beltline welds.

## 8. SAFTEY AND MITIGATION

Mitigation strategies for PTS have been considered for years. A direct strategy is to avoid the neutron embrittlement. This has been accomplished with neutron shielding and spot fluence reduction at key welds. In addition, core loading patterns can be setup to minimize neutron leakage from the core.

There are system and thermal-hydraulic considerations that can be accomplished also. For instance, in cold weather plants, the safety injection water could be heated to mitigate the cooldown scenario. There are also plant control system changes that have been considered. Tripping feedwater pumps on high SG levels, isolation of steam generators during MSLB and reducing HPI flows during small LOCAs are just a few issues that can be considered. The first consideration during a nuclear accident is keeping the reactor system cool and filled with water. In some plants, these considerations have to be carefully considered (and not over done) in light of the PTS considerations.

## 9. SUMMARY

PTS is a multi-disciplined problem that challenges the RPV itself. Assurance of the integrity of this critical component of the nuclear power plant system warrants a careful consideration of the PTS issue. PTS is a plant specific problem and many of the parameters that govern the PTS evaluations are plant specific. Ultimately, the embrittlement of the reactor vessel determines the significance of PTS for a particular plant.

The tools available to analyses PTS scenarios continue to evolve. Thermal-hydraulic system codes are becoming more robust and more efficient to run which allows the analyst to consider a large number of overcooling scenarios.

For a detailed understanding of the flow and heat transfer behavior in the downcomer region, there is a need for adequate experimental results to develop correlations and validate modeling approaches. CFD techniques are showing great promise in this area and much work is being accomplished to develop these tools (specifically in the area of multi-phase flow and direct contact condensation). The coupling of CFD techniques into the system code evaluation models should provide a more reliable solution in the future. In addition, CFD tools are commonly coupled with stress analysis (and fracture options) tools which provides a robust method to compute the complete behavior in the downcomer region.

As our understanding and capabilities continue to improve, the nuclear safety community will benefit from updated and specific best practice guidelines for all aspects for the PTS analysis. An update, 10+ years later, of the PTS ICAS is something to consider.

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